ACCESSION #: 9603120224

LICENSEE EVENT REPORT (LER)

FACILITY NAME: TURKEY POINT UNIT 3 PAGE: 1 OF 4

DOCKET NUMBER: 05000250

TITLE: AUTOMATIC TURBINE TRIP/REACTOR TRIP DUE TO HIGH STEAM

GENERATOR LEVEL

EVENT DATE: 02/09/96 LER #: 96-002-00 REPORT DATE: 03/06/96

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 60

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: J.A. Hickey, Licensing Engineer TELEPHONE: (305) 246-6668

COMPONENT FAILURE DESCRIPTION:

CAUSE: B SYSTEM: BA COMPONENT: 65 MANUFACTURER: T147

B SJ V C255

REPORTABLE NPRDS: Y

Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On February 09, 1996, Florida Power & Light Company's Turkey Point Unit 3 was operating in mode 1 at 60% power to support condenser waterbox cleaning.

At 2329 the "B" Steam Generator Feed Pump (SGFP) was stopped to monitor its discharge check valve closing stroke. The discharge check valve did not stroke closed as

expected. At 2334 the resulting feed flow transient caused the "C" Steam Generator (S/G) level to increase, resulting in a turbine trip. A reactor trip by turbine trip occurred immediately thereafter.

The cause of the turbine trip/reactor trip was cognitive personnel error. The operator failed to effectively control the "C" S/G level during the feed flow transient.

The NRC operations center was notified at 0035 on February 10, 1996, in accordance with 10 CFR 50.72 (b) (2) (ii), Reactor Protection System Actuation.

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I. DESCRIPTION OF THE EVENT

On February 09, 1996, Florida Power & Light Company's Turkey Point Unit 3 was operating in mode 1 at 60% power to support condenser waterbox cleaning.

An investigation into the source of a suspected loose part in the 6B High Pressure Feedwater Heater (SN:HX) was in progress. monitoring the performance of each SGFP discharge check valve (SJ:V) was part of the investigation. A pre-evolution briefing for cycling the SGFPs was conducted by on-shift supervision and a management designee. Expected plant response and potential problems were discussed, including the failure of the discharge check valve to close.

The "A" SGFP (SK:P) was cycled with no abnormal indications. The diagnostic test equipment was transferred to the "B" SGFP discharge check valve and another control room pre-evolution briefing was held. At 2329 the "B" SGFP was secured. The "B" SGFP discharge check valve did not immediately close. Reverse feed flow back through the open "B" SGFP discharge check valve caused feed flow on all three S/G's to drop to approximately 1/2 of the original flow. The operators took manual control of all three S/G Feedwater Regulating Valves (FRV) (SJ:FCV) and stabilized levels at or above approximately 38%. During this time the "B" SGFP discharge check valve slowly drifted closed and the "B" SGFP discharge MOV (SJ:ISV) closed automatically as expected.

During the subsequent recovery the operator stated the "C" FRV was placed in automatic with steam flows and feed flows matched and level at program, (approximately 60%). The "A" and "B" S/G levels continued to rise above program. The operator manually reduced feed flow to stop the level increases. This action successfully halted the level increase in the "A" and "B" S/G's. Because the "C" S/G FRV was believed to be in automatic, crew attention was focused on the "A" and "B" S/G levels. The manual decrease in feed flow for the "A" and "B" S/G's forced more flow to the "C" S/G, resulting in a level increase. The "C" S/G level was approximately 756 when the operator took manual control of the "C" FRV in an attempt to lower level. The level increase could not be stopped and direction was given for a manual reactor trip. The manual action could not be completed before reaching the Hi-Hi S/G level turbine trip setpoint of 80%.

The reactor was tripped by the turbine at 2334 as expected. The Hi-Hi S/G level trip is also a Feedwater Isolation Signal which results in an Auxiliary Feedwater (AFW) System automatic start. At 2350, while reducing Train 1 AFW flows, the "A" AFW pump (BA:P) tripped on electronic and mechanical overspeed.

Unit 3 was stabilized at no-load conditions and the Emergency Operating Procedures were exited approximately 15 minutes later at 0005 on February 10, 1996.

At 0200 on February 10, 1996, the "C" AFW Pump was realigned to restore Train 1 AFW System operability.

II. CAUSE OF THE EVENT

Root Cause

The root cause of the event was cognitive personnel error. The "C" S/G FRV was placed in automatic control and acceptable controlling performance was not verified. With the "C" FRV in automatic, the operator's attention was diverted to other plant parameters. Required operator actions to compensate for the subsequent increase in "C" S/G level did not occur.

Contributing Cause

The "B" SGFP discharge check valve failed to close as expected. The cause of the slow closure, was the failure of tack welds on one of the hinge pin retaining bolts. The retaining bolt unscrewed, which allowed the hinge pin to fall out.

"A" AFW Pump Overspeed

The "A" AFW Pump tripped on electronic and mechanical overspeed due to binding of the governor. The cause of the binding was pitting on the governor stem, induced by environmental corrosion.

III. ANALYSIS OF THE EVENT

The Updated Final Safety Analysis Report (UFSAR) analysis assumes a loss of normal feedwater to all steam generators due to the loss of the feedwater pumps or valve malfunction. In the February 09 event, feed flow was not lost completely, but was significantly reduced. All steam generators were initially affected by the reduction in feedwater flow.

In the analysis, the reactor trip is expected to occur due to a Low-Low Level in any steam generator or Steam/Feedwater Flow Mismatch Coincident with Low Level in any steam generator. In this event neither trip occurred due to operator actions which stabilized the S/G levels above the Low and Low-Low Level setpoints. The analysis shows that following a loss of normal feedwater, AFW is capable of removing the stored and residual heat, thus preventing either overpressurization of the reactor coolant system or loss of water from the reactor core. The analysis also assumes only one AFW pump is available due to a single failure. Two AFW pumps were available following the overspeed of the "A" AFW pump, therefore, the plant's response was bounded by the analysis. This event did not compromise the health or safety of plant personnel or the general public.

This event is reportable under the requirements of 10 CFR 50.73(a)(2)(iv).

IV. CORRECTIVE ACTIONS

1. Both hinge pins on the "B" SGFP discharge check valve have been replaced. The new tack welds on the retaining bolts have been verified as adequately sized. The "A" SGFP discharge check valve was inspected and reassembled, the retaining bolt tack welds have been verified as adequately sized. All similar check valves regardless of application have been or will be inspected for adequately sized retaining bolt tack welds.

2. Remedial simulator training was conducted for the individual controlling the "C" S/G level. The conditions of the event were approximately duplicated and the individual successfully controlled the S/G level.

3. The governor stem on the "A" AFW Pump has been replaced. A special surveillance for stroking the governor stems is in place. When the AFW Pump governor stems are replaced with an upgraded corrosion resistant stem, the special surveillance will be discontinued.

4. The "C" S/G Flow Control Valve performance in automatic was investigated. The results of the investigation revealed no abnormalities with the "C" S/G flow control valve.

5. Operations management will review this event with each operating crew.

a. The Nuclear Plant Supervisor involved in this event has completed a Command and Control performance evaluation of the event.

6. Training will review/establish scenarios which approximate this event.

7. A Post-Trip meeting was held and video taped. In attendance were approximately 65 individuals from Operations,Training, Site Management and the President of the Nuclear Division. The meeting format was an "open forum" to ensure all parties understood the event. The meeting provided a vehicle to clearly reenforce Management's expectation that prompt manual action, including a manual reactor trip must be taken prior to challenging any automatic plant protective feature.

V. ADDITIONAL INFORMATION

A similar SGFP discharge check valve failure occurred on Unit 4 in 1993.

A turbine trip/reactor trip on Hi-Hi S/G level occurred on Unit 4 in 1993. The cause was improperly valving in a high pressure feedwater heater.

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MAR 06 1996

FPL

L-96-043

10 CFR 50.73

U.S. Nuclear Regulatory Commission

Attn: Document Control Desk

Washington, D. C. 20555

Gentlemen:

Re: Turkey Point Unit 3

Docket No. 50-250

Reportable Event: 96-002-00

Automatic Turbine Trip/Reactor Trip due to High Steam Generator Level

The attached Licensee Event Report, 250/96-002-00, is being provided in

accordance with 10 CFR 50.73(a)(2)(iv).

Should there be any questions, please contact us.

Very truly yours,

Robert J. Hovey

Vice President

Turkey Point Plant

JAH

attachment

cc: S. D. Ebneter, Regional Administrator, Region II, USNRC

T. P. Johnson, Senior Resident Inspector, USNRC,

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